Nuclear Energy University Programs (NEUP) Fiscal Year (FY) 2015 Annual Planning Webinar

Advanced Structural Materials (RC-3)

William Corwin
Office of Advanced Reactor Technologies
U.S. Department of Energy

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Structural Materials Are Critical for Advanced Nuclear Reactor Technologies

- Development and qualification of advanced structural materials are critical to the design and deployment of the advanced nuclear reactor systems that DOE is developing
  - High and Very High Temperature Gas Cooled Reactors (HTGRs and VHTRs)
  - Sodium Cooled Fast Reactors (SFRs)
  - Fluoride Salt Cooled High Temperature Reactors (FHRs)
- Structural materials must perform over design lifetimes for pressure boundaries, reactor internals, heat transfer components, etc.
- Performance of metallic alloys for the long times and high operating temperatures required is being examined under the Advanced Reactor Technologies (ART) Program

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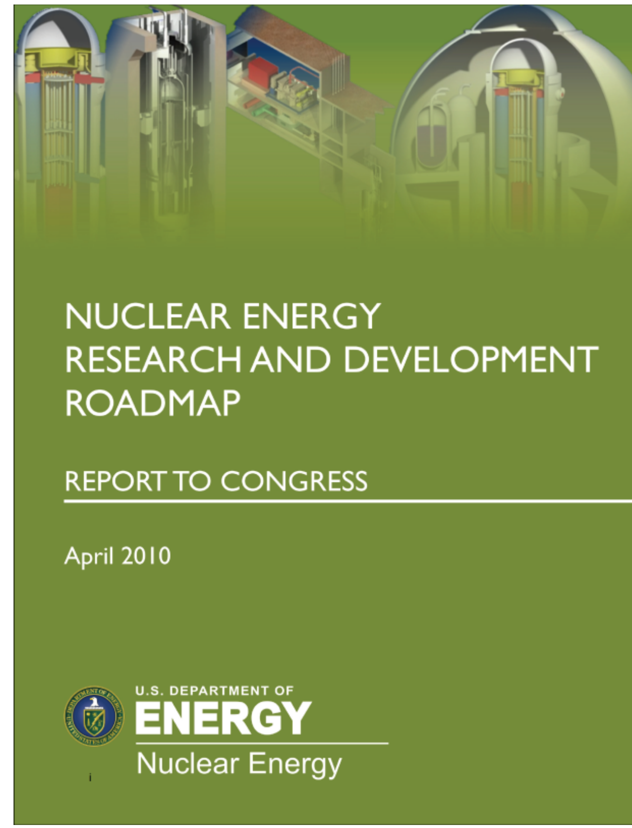
ART Program Supports Fast Reactor Development

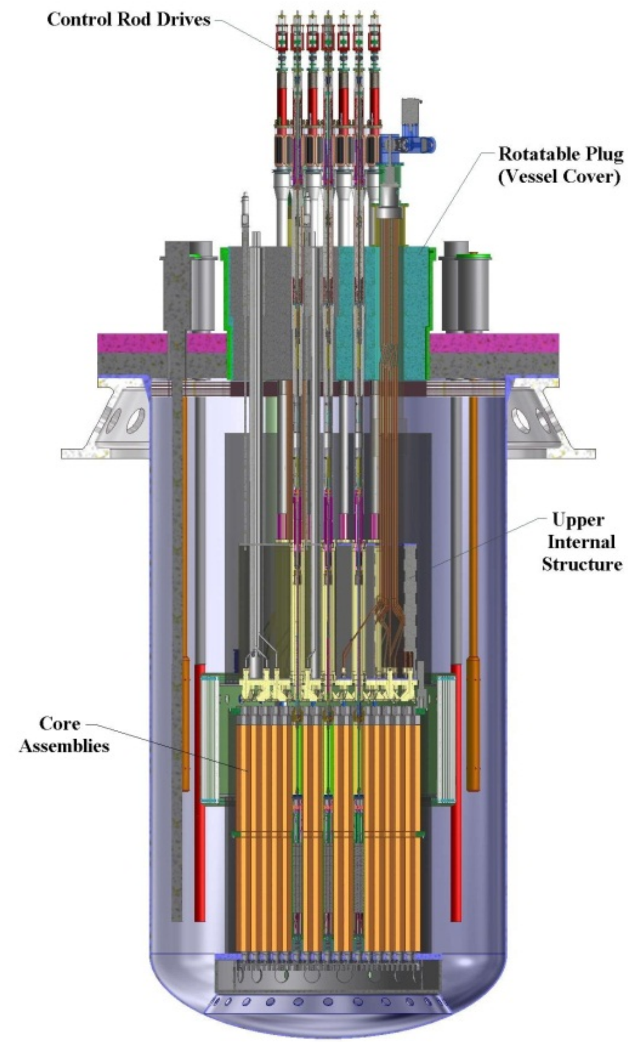
- **Specific Objective - Develop advanced fast reactor technology solutions to allow commercial deployment by 2050 timeframe**
- **Supports multiple high-level Objectives identified in the 2010 Nuclear Energy R&D Roadmap (2 & 3)**
  - (2) Develop improvements in the affordability of new reactors to enable nuclear energy to help meet the Administration’s energy security and climate change goals
  - (3) Develop sustainable nuclear fuel cycles

    “The overall goal is to have demonstrated the technologies necessary to allow commercial deployment of solution(s) for the sustainable management of used nuclear fuel that is safe, economic, and secure and widely acceptable to American society by 2050.”



ART Program Includes Advanced Materials R&D Activities

- Development and qualification of improved ferritic-martensitic steels and austenitic alloys for fast reactor systems
- Advanced Fast Reactor-100 is an example of fast reactor systems
  - Targets local small grids with limited needs for on-site refueling
  - Reactor power of 250MWt/100MWe, core life (30 years), plant life (60 years)
  - Transportable from pre-licensed factory
  - Adopts innovative technologies and advanced materials

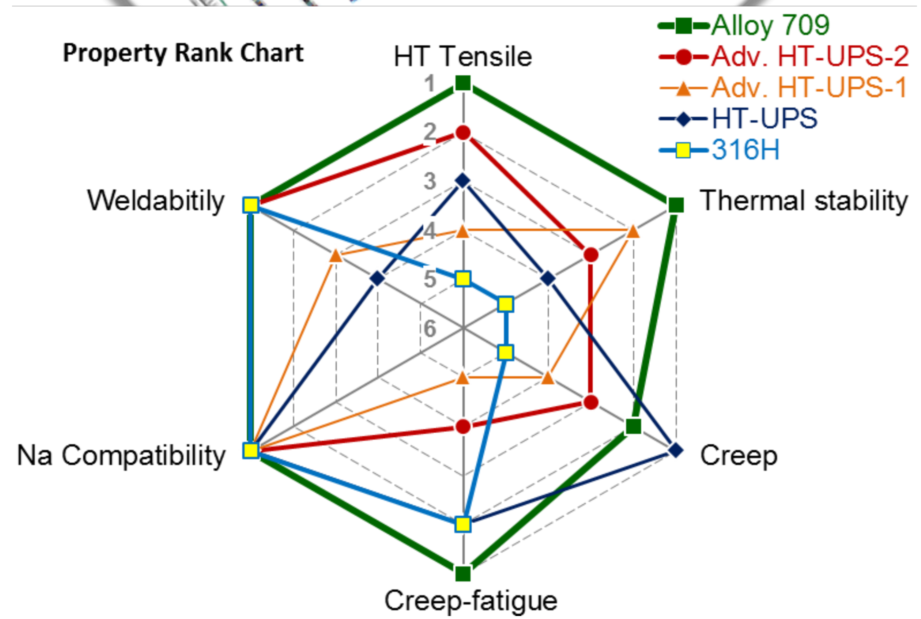


Austenitic Alloy Down-Selection Has Been Completed


Candidate Developmental Alloys for Improved Structural Materials for Advanced Fast Reactors
March 2008

Goal is to provide structural materials for advanced reactors (esp. for fast reactors) with better high temperature strength to increase thermal efficiency and provide improved design & safety margins

- **Extensive selection of material classes for further development (FY 2008)**
- **Austenitic alloys considered during down-selection (completed in FY 2012):**
  - **HT-UPS (Fe-14Cr-16Ni base): Good creep & sodium compatibility, poor weldability**
  - **Advanced HT-UPS (Fe-13Cr-16Ni base): Improved weldability, moderate creep**
  - **Alloy 709 (Fe-20Cr-25Ni base): Has optimum balanced properties**
  - **316H (Fe-18Cr-12Ni base): Reference material**



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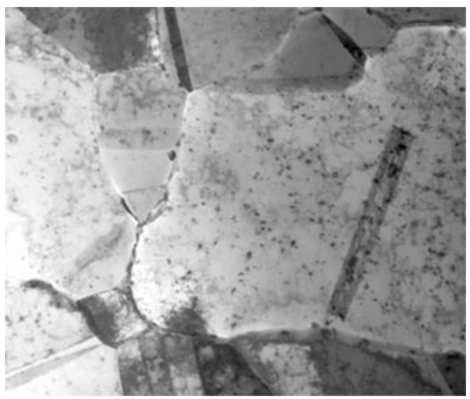
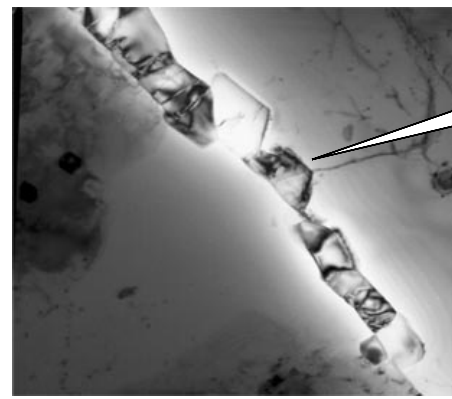
Alloy 709 Is Nitrogen-Stabilized, Niobium-Strengthened Austenitic Alloy with ≈ 2X Creep Strength of 316 SS

- Based on the Fe-20Cr-25Ni composition for power boiler applications
- Excellent oxidation resistance at high temperatures
- Niobium and nitrogen additions increase the tensile and creep strength

Microstructure after creep test for 5015h @ 750C/100MPa (TEM-BF images)

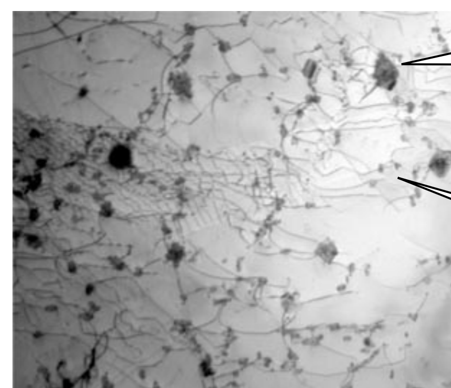


GB



$M_{23}C_6$+M_6C

GI

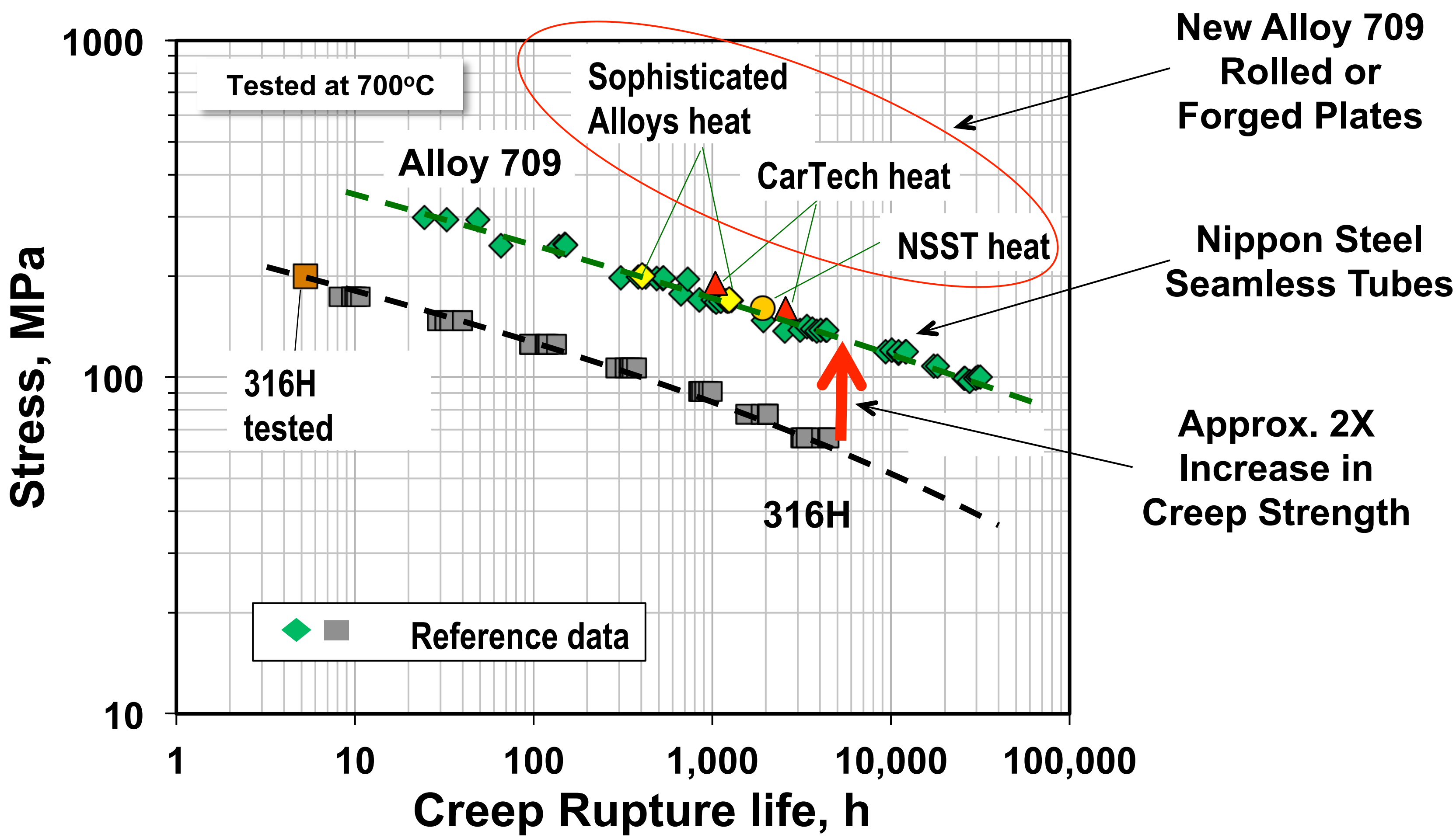


$M_{23}C_6$ & NbC

NbC

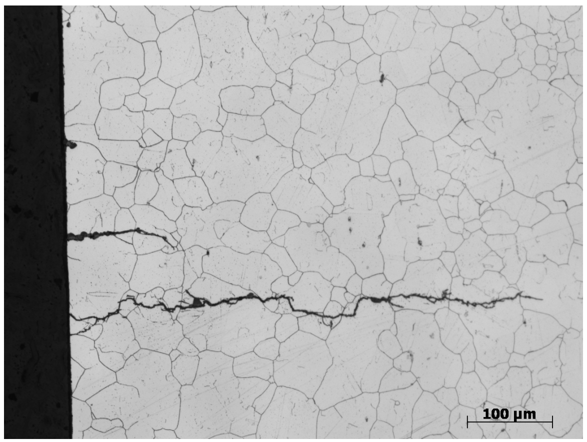
- Most of the grain boundary (GB) is fully covered by precipitates
- Two different sizes of precipitates in the grain interior (GI)

Alloy 709 Has Enhanced High Temperature Mechanical Properties vs Reference 316H Stainless Steel



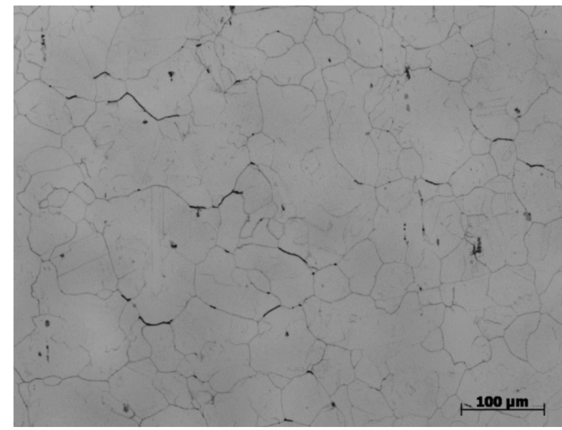
Fatigue & Creep-Fatigue Are Dominant Failure Mechanisms Over Reactor Component Operating Lifetimes

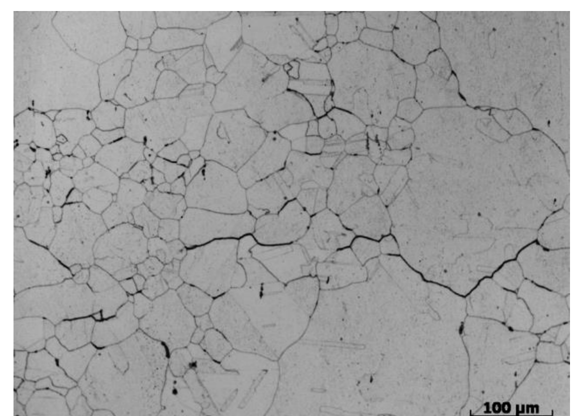
Fatigue loading at 650C,1.0% total strain range, R-ratio = -1



Transgranular cracking initiated from surface for fatigue loading

Creep-fatigue loading at 650C,1.0% total strain range, and a 30 min. tensile hold, R-ratio = -1



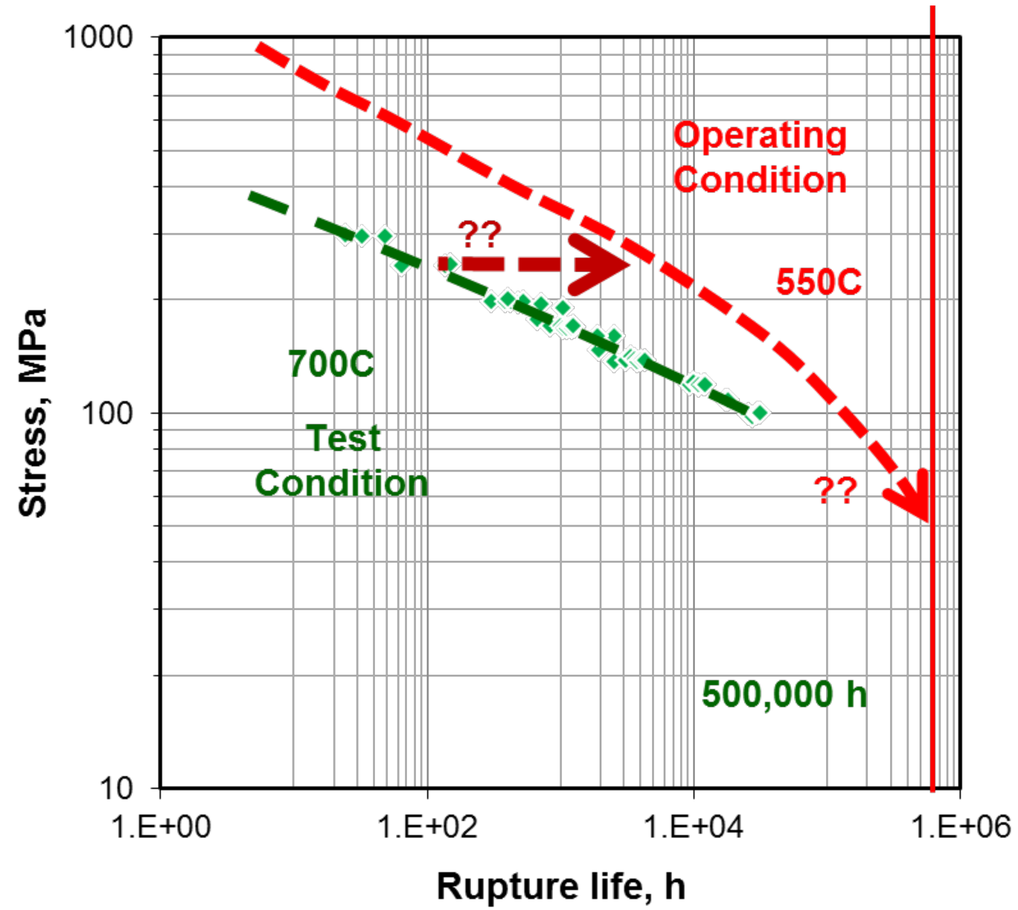


Interior grain boundary cracking for creep-fatigue

Alloy 709

RC-3.1 Structural Materials R&D Required to Support the Design of Fast Reactors for a 60-Year Plant Life

- ASME Code design allowable stresses depend on design lifetime. For 60-year plant life need
  - Stress to cause rupture in 500,000 h
  - Stress to cause initiation of tertiary creep in 500,000 h
  - Stress to cause 1% strain in 500,000 h
- Data extrapolation from test conditions to operating condition is necessary
- Understanding of creep & creep-fatigue deformation, microstructural evolution & damage (GB cavitation) mechanisms will provide guidance for design data extrapolation



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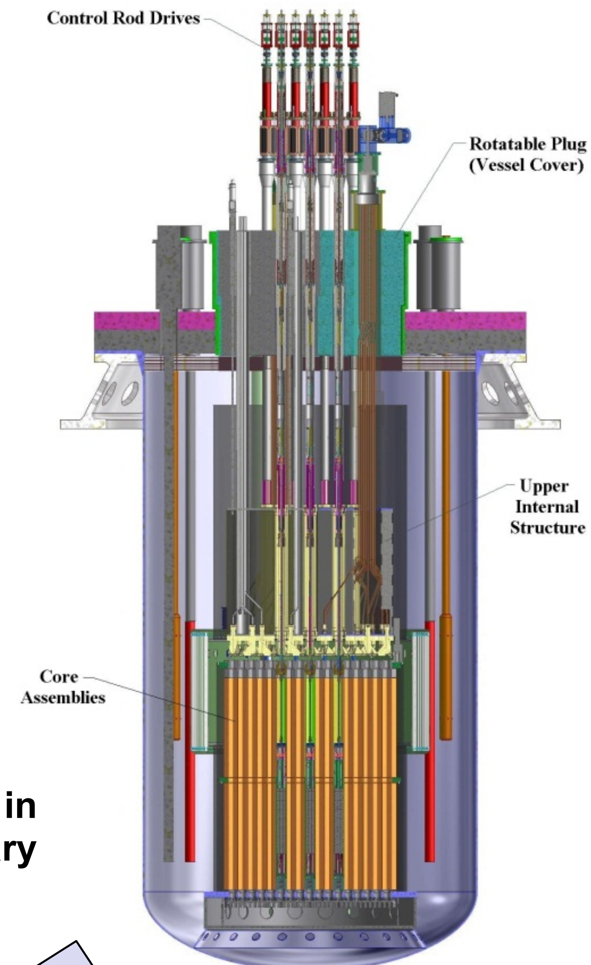
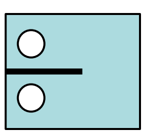
RC-3.2 Structural Materials R&D Required to Support the Operations of Fast Reactors for 60-Year Plant Life

- Flaw evaluation procedure is required for the disposition of detected indications/flaws during in-service inspection
- It is also required to develop schedule for in-service inspection for safe operation of the reactor plant
- Extrapolation of subcritical creep and creep-fatigue crack growth data from accelerated test condition to operating condition is necessary
- Understanding of subcritical creep and creep-fatigue crack growth mechanisms is important in providing guidance for the establishment of accelerated testing conditions and for test data extrapolation to operating condition



Laboratory compact tension specimen



Subcritical flaws in pressure boundary components

ART NEUP Structural Materials Research Supports Closely Related Needs for Code Qualification of Alloy 709 in FY15

- **RC-3.1 Creep and creep-fatigue deformation and grain boundary cavitation mechanisms for Alloy 709**
  - The expected outcome of selected projects is to provide a mechanistic-based, validated approach to extrapolate creep and creep-fatigue data from accelerated test conditions to fast reactor operating conditions in support of ASME Code stress allowable development

- **RC-3.2 Creep and creep-fatigue crack growth mechanisms for Alloy 709**
  - The expected outcome of selected projects is to provide validated approaches to subcritical crack growth data extrapolation and flaw evaluation in support of sodium fast reactor operations

RC-3.1 Creep & Creep-Fatigue Deformation and Grain Boundary Cavitation Mechanisms in Alloy 709

- **Proposals are sought to develop fundamental understanding of deformation mechanisms (e.g. dislocation or diffusion controlled creep), microstructural evolution, and damage mechanisms (e.g. grain boundary cavitation) of Alloy 709 at elevated temperatures under creep and creep-fatigue loading conditions to enable intelligently applying and interpreting data from accelerated testing**

- **The proposed research should focus on**
  - **Similarities or differences of these characteristics between the accelerated test conditions and fast reactor operating conditions**
  - **Data extrapolation strategies**
  - **Approaches might include novel experimental and characterization methods**

RC-3.2 Creep and Creep-Fatigue Crack Growth Mechanisms for Alloy 709

- Proposals are sought to develop fundamental understanding of subcritical crack growth mechanisms of Alloy 709 at elevated temperatures under creep and creep-fatigue loading conditions to enable intelligently applying and interpreting data from accelerated testing
- The proposed research should focus on
  - Similarities or differences of creep and creep-fatigue subcritical crack growth mechanisms between the accelerated test conditions and fast reactor operating conditions, and between as-received materials and materials that have undergone prior service conditions
  - Extrapolation strategy for subcritical crack growth data obtained from fracture specimens
  - Approaches might include novel experimental and characterization methods, and validated engineering flaw evaluation analysis techniques